JAN 1 5 1976

Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
ATTN: Mr. I. S. Mitchell, III
Vice President and Secretary
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant Unit 1. This consists of a change to the Technical Specifications, Appendix A, and is in response to your request dated November 29, 1974.

The amendment incorporates into the Edwin I. Hatch Nuclear Plant Unit 1 Technical Specifications changes to the Administrative Controls. Changes to your proposal were necessary to meet our requirements. These have been discussed with your staff. The Technical Specifications are based on the regulatory positions described in Guides 1.8, "Personnel Selection and Training", 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, and 1.33, "Quality Assurance Program Requirements".

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest". Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File" of which you were previously provided a copy. This report is modified by updated instructions dated December 8, 1975 which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations", a copy of which is also enclosed. This guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your Technical Specifications. Reports that are required by the regulations have not been repeated in your Technical Specifications.

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Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Please note that we have discontinued the use of separate identifying numbers for changes to technical specifications. Sequential amendment numbers will be continued as in the past.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 18
2. Regulatory Guide 1.16
3. Updated Instructions
4. Regulatory Guide 10.1
5. Safety Evaluation
6. Federal Register Notice

cc: w/enclosures
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. DPR-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Georgia Power Company and Oglethorpe Electric Membership Corporation (the licensees) dated November 29, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

"2. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised".

3. This license amendment is effective 60 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: January 15, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 18

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Remove pages 1.0-1, 1.0-7, 3.8-3, 6.0-1 thru 6.0-25, Figure 6.0-1 and Figure 6.0-2 and replace with the attached revised pages 1.0-1, 1.0-7, 3.8-3, 6-1 thru 6-24, Figure 6.2-1 and Figure 6.2-2. No changes have been made on page 1.0-2.

1.0 Definitions

The following terms are defined so that a uniform interpretation of these specifications may be achieved.

B. Cold Shutdown Condition - Cold shutdown condition means reactor operation with the Mode Switch in the SHUTDOWN position, coolant temperature less than or equal to 212°F, and with no core alterations permitted.

C. Core Alteration - Core alteration is the act of moving any component in the region above the core support plate, below the top guide and within the core shroud. Normal control rod movement with the control rod drive hydraulic system or normal movement of in-core instrumentation is not considered a core alteration.

D. Design Power - Design power refers to the power level at which the reactor is producing 105 percent of reactor vessel rated steam flow. Design power does not necessarily correspond to 105 percent of rated reactor power. The stated design power in megawatts thermal (MWt) is the result of a heat balance for a particular plant design. For Hatch Nuclear Plant Unit 1 the design power is 2537 MWt. Design power is used as an initial condition in transient and accident analyses.

E. Engineered Safety Features - Engineered safety features are those features provided for mitigating the consequences of postulated accidents, including for example containment, emergency core cooling, and standby gas treatment system.

F. Hot Shutdown Condition - Hot shutdown condition means reactor operation with the Mode Switch in the SHUTDOWN position, coolant temperature greater than 212°F, and no core alterations are permitted.

G. Hot Standby Condition - Hot standby condition means reactor operation with the Mode Switch in the START & HOT STANDBY position, coolant temperature greater than 212°F, reactor pressure less than 1045 psig, critical.

H. Immediate - Immediate means that the required action shall be initiated as soon as practicable, considering the safe operation of the Unit and the importance of the required action.

I. Instrument Calibration - An instrument calibration means the adjustment of an instrument output signal so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors.

J. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

MM. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ratio (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

NN. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

OO. Cumulative Downtime - The cumulative downtime for those safety components and systems whose downtime is limited to 7 consecutive days prior to requiring reactor shutdown shall be limited to any 7 days in a consecutive 30 day period.

3.8A Miscellaneous Radioactive Material Sources

The objective of this specification is to assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits. The result of these tests are to be recorded in accordance with Specification 6.10.1.h.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Superintendent shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

Offsite

6.2.1 The offsite organization for plant management and technical support is shown in relation to plant supervision on Figure 6.2-1.

Plant Staff

6.2.2 The plant organization shall be as shown on Figure 6.2-2, and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shut-down, and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

6.3 PLANT STAFF QUALIFICATIONS

The qualifications with regard to educational and experience backgrounds of key supervisory and professional personnel at the time of initial core loading or appointment to the active position shall be as follows:

6.3.1 Plant Superintendent and Assistant Plant Superintendent

The Plant Superintendent and Assistant Plant Superintendent shall have ten years of responsible power plant experience, of which a minimum of two years shall be nuclear power plant experience. A maximum of four years of the remaining eight years of experience may be fulfilled by academic training on a one-for-one time basis. This academic training shall be in an engineering or scientific field generally associated with power production. The Plant Superintendent and Assistant Plant Superintendent shall have acquired the experience and training normally required for examination by the AEC for a Senior Reactor Operator License whether or not the examination is taken. The Plant Superintendent or the Assistant Plant Superintendent shall obtain a Senior Reactor Operator License.

6.3.2 Operations Supervisor

The Operations Supervisor shall have a minimum of eight years of responsible power plant experience, of which a minimum of three years shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He shall hold a Senior Reactor Operator License.

6.3.3 Assistant Operations Supervisor

The Assistant Operations Supervisor shall have a minimum of a high school diploma or equivalent and four years of responsible power plant experience, of which a minimum of one year shall be nuclear power plant experience. A maximum of two years of the remaining three years of power plant experience may be fulfilled by academic or related technical training on a one-for-one time basis. He shall hold a Senior Reactor Operator License.

6.3.4 Technical Supervisor

The Technical Supervisor shall have a minimum of eight years in responsible positions, of which one year shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience should be fulfilled by satisfactory completion of academic training. He shall hold a Senior Reactor Operator License.

6.3.5 Maintenance Supervisor

The Maintenance Supervisor shall have a minimum of seven years of responsible power plant experience or applicable industrial experience, a minimum of one year of which shall be nuclear power plant experience. A maximum of two years of the remaining six years of power plant or industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He should have non-destructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

6.3.6 Health Physics Radiochemist

The Health Physics Radiochemist shall have an associate engineering (or equivalent) degree and five years experience in chemistry and radiation protection. A minimum of two years of this five years experience should be related technical training. A maximum of four years of the five years experience may be fulfilled by related technical or academic training.

6.3.7 Test Engineer

The Test Engineer shall have a minimum of five years experience in instrumentation and control, of which a minimum of six months shall be in nuclear instrumentation and control. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

6.3.8 Reactor Engineer

The Reactor Engineer shall have a minimum of a Bachelor's Degree in Engineering or the physical sciences, and two years experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing programs.

6.3.9 Site Quality Assurance Representative

The Site Quality Assurance Representative shall have a four year degree or equivalent with experience or training in plant operations.

6.3.10 Quality Control Engineer

The Quality Control Engineer shall have an engineering (or equivalent) degree with experience in power plant operation.

6.3.11 Training Specialist

The Training Specialist shall have an engineering (or equivalent) degree with experience in reactor operations.

6.3.12 Shift Supervisors

The Shift Supervisors shall have a minimum of a high school diploma or equivalent and four years of responsible power plant experience, of which a minimum of one year shall be nuclear power plant experience. A maximum of two years of the remaining three years of power plant experience may be fulfilled by academic or related technical training on a one-for-one time basis. Each shall hold a Senior Reactor Operator License.

6.3.13 Plant Operators and Assistant Plant Operators

The Plant Operators and Assistant Plant Operators shall have a high school diploma or equivalent and two years of power plant experience, of which a minimum of one year shall be nuclear power plant experience. Each shall hold a Reactor Operator License.

6.3.14 Auxiliary Equipment Operators

The Auxiliary Equipment Operators shall have a high school diploma or equivalent, and should possess a high degree of manual dexterity and mature judgement.

6.3.15 Technicians

Technicians in responsible positions shall have a minimum of two years of working experience in their specialty and should have a minimum of one year of related technical training in addition to their experience.

6.3.16 Maintenance Personnel

Maintenance personnel in responsible positions shall have a minimum of three years experience in one or more crafts. Any time maintenance is performed on a safety related system, those maintenance personnel performing the task shall meet this requirement or they shall receive detailed supervision from one who does. They should possess a high degree of manual dexterity and ability and should be capable of learning and applying basic skills in maintenance operations.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Training Specialist and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971, and Appendix "A" of 10 CFR, part 55.

6.5 REVIEW AND AUDIT

6.5.1 Plant Review Board (PRB)

Function

6.5.1.1 The Plant Review Board shall function to advise the Plant Superintendent on all matters related to nuclear safety.

Composition

6.5.1.2 The Plant Review Board as a minimum shall be composed of the:

Chairman - Plant Superintendent
Vice Chairman - Assistant Plant Superintendent
Member - Operations Supervisor
Member - Technical Supervisor
Member - Maintenance Supervisor
Member - Health Physics Radiochemist
Member - Quality Control Engineer

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the Plant Superintendent to serve on a temporary basis; however, no more than two alternates shall participate in PRB activities at any one time.

Meeting Frequency

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman or as requested by a PRB member.

Quorum

6.5.1.5 A quorum of the PRB shall consist of four members, one of which shall be the Chairman or Vice Chairman.

Responsibilities

6.5.1.6 The Plant Review Board shall be responsible for:

a. Review of (1) all procedures required by Specification 6.8 and changes thereto, and (2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

Responsibilities (continued)

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Manager of Production and to the Chairman of the Safety Review Board.

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Safety Review Board.

h. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Safety Review Board.

i. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Safety Review Board.

Authority

6.5.1.7 The Plant Review Board shall:

a. Recommend to the Plant Superintendent written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide immediate written notification to the Manager of Production and to the Chairman of the Safety Review Board of disagreement between the PRB and the Plant Superintendent; however, the Plant Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

Records

6.5.1.8 The Plant Review Board shall maintain written minutes of each meeting and copies shall be provided to the Manager of Production and to the Chairman of the Safety Review Board.

6.5.2 Safety Review Board (SRB)

6.5.2.1 The Safety Review Board shall function to provide independent review and audit and have the competence required to review problems in the following areas:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Quality assurance practices
i. (Other appropriate fields associated with the unique characteristics of the nuclear power plant.)

Composition

6.5.2.2 The SRB shall consist of a minimum of seven persons who as a group collectively provide expertise in the operation of a nuclear power plant. The Chairman and Vice Chairman and other members shall be appointed by the Company Senior Vice President - Power Supply or such other person as he shall designate. The composition of the SRB shall meet the intent of ANSI N18.7-1972.

Alternates

6.5.2.3 Alternate representatives, who will not have voting privileges, may be designated on a temporary basis by the member being absent. No more than two alternates shall participate in SRB activities at any one time.

Consultants

6.5.2.4 Consultants shall be utilized as determined by the SRB Chairman or Vice Chairman to provide expert advice to the SRB.

Meeting Frequency

6.5.2.5 The SRB shall meet at least once per calendar quarter during the initial year of plant operation following fuel loading and at least once per six months thereafter.

Quorum

6.5.2.6 A quorum of SRB shall consist of four regular members, one of whom shall be the Chairman or Vice Chairman. No more than two of the quorum shall have line responsibility for operation of the facility.

Review

6.5.2.7 The SRB shall review:

a. The safety evaluations for (1) changes to procedures, equipment or systems, and (2) tests or experiments

Review (continued)

completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All events which are required by regulations or Technical Specifications to be reported to the NRC in writing within 24 hours.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meeting minutes of the Plant Review Board.

Audits

6.5.2.8 Audits of plant activities shall be performed under the cognizance of the SRB. These audits shall encompass:

a. The conformance of plant operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training, and qualifications of the entire plant staff at least once per year.

c. The results of all actions taken to correct deficiencies occurring in plant equipment, structures, systems, or method of operation that affect nuclear safety at least once per six months.

Audits (continued)

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The plant Emergency Plan and implementing procedures at least once per two years.

f. The plant Security Plan and implementing procedures at least once per two years.

g. Any other area of plant operation considered appropriate by the SRB or the Senior Vice President - Power Supply.

Authority

6.5.2.9 The SRB shall report to and advise the Senior Vice President Power Supply on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

Records

6.5.2.10 Records of SRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each SRB meeting shall be prepared, approved, and forwarded to the Senior Vice President - Power Supply within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7. e, f, g, and h above shall be prepared. approved, and forwarded to the Senior Vice President - Power Supply within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Vice President - Power Supply and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of an REPORTABLE OCCURRENCE:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each Reportable Occurrence Report submitted to the Commission shall be reviewed by the PRB and submitted to the SRB and the Manager of Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. If a Safety Limit is exceeded, the reactor shall be shutdown and reactor operation shall not be resumed until authorized by the NRC.

b. The Safety Limit violation shall be reported to the Commission, the Manager of Production and to the Chairman of the SRB immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon plant components, systems or structures, and (3) corrective action taken to prevent re-currence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the SRB, and the Manager of Production within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented, and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972. and Appendix "A" of of Regulatory Guide 1.33, November, 1972, except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PRB and approved by the Plant Superintendent prior to implementation and periodically as set forth in each document.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made, provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

6.8 PROCEDURES (Continued)

c. The change is documented, reviewed by the PRB, and approved by the Plant Superintendent on a timely basis.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulation, Chapter I, the following identified reports shall be submitted to the Director of Inspection and Enforcement Regional Office II, Atlanta, Georgia 30303.

6.9.1 Routine Reports

a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Operating Report. Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality. The annual report shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(1) A narrative summary of operating experience during the report period relating to safe operation of the facility, including

safety-related maintenance not covered in item 1.b.(2)(e) below.

(2) For each outage or forced reduction in power[1/] of over 20% of design power level where the reduction extends for greater than four hours:

(a) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

(b) a brief discussion of (or reference to reports of) any abnormal occurrences pertaining to the outage or power reduction;

(c) corrective action taken to reduce the probability of recurrence, if appropriate;

(d) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,[2/] use the generator off-line hours; for forced reduction in power, use the approximate duration of operation at reudced power);

(e) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

(f) a report of any single release of radioactivity or radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

(3) A tabulation (supplementing the requirements of Section 20.407 of 10 CFR Part 20) on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job function, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whoe body dose received from external sources shall be assigned to specific major work functions.

1/The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this action.

2/The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

(4) A report of fuel performance, as follows:

(a) Reporting requirements as specified in 4.6.F.1.

(b) All findings from failed fuel examinations, including results of eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience should be submitted on a monthly basis. The completed reports should be sent to the Director, Office of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office II, Office of Inspection and Enforcement no later than the tenth of each month following the calendar month covered by the report.

6.9.1.2 Reportable Occurrences

Reportable Occurrences, including corrective actions and measures to prevent reoccurrences, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification With Written Followup. The types of events listed below should be reported as expeditiously as possible, but within 24 hours, by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Region II Office of Inspection and Enforcement, or his designate, no later than the first working day following the event, with a written followup report within two weeks. The written report should include, as a minimum, a completed copy of the licensee event report form. Information provided on the licensee event report form should be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system or other systems subject to limiting safety-system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety-system setting in the technical specifications or failure to complete the required protective function.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

NOTE: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 6.9.1.2.b.(2).

(3) Abnormal degradation in fuel cladding, reactor coolant pressure boundary, or primary containment.

(4) Reactivity Anomalies

(a) Discovery of disagreement with predicted value of reactivity balance greater than or equal to 1% $\Delta k/k$.

(b) A projection of a reactivity balance that would threaten the ability to attain required shutdown margin.

(c) Short-term reactivity increases that correspond to a reactor period of less than 5 seconds, or if subcritical an unplanned reactivity insertion of more than .5% $\Delta k/k$.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to function to cope with accidents analyzed in the FSAR.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to function to cope with accidents analyzed in the FSAR.

(7) Conditions arising from natural or man-made events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

(8) Errors in the transient or accident analyses or in the methods used for such analyses that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

(9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in

the safety analysis report or technical specifications (including bases) or discovery during plant life of conditions that were specifically considered in the safety analysis report or technical specifications that require remedial action or correction measures to prevent the existence or development of an unsafe condition.

b. Thirty Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the Region II, Office of Inspection and Enforcement no later than 30 days following the event. The written report shall include, as a minimum, a completed copy of a licensee event form. Information provided on the licensee event form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems, other than those specified in 6.9.1.2.(3), designed to contain radioactive material resulting from the fission process.

6.9.1.3 Unique Reporting Requirements

Special reports shall be submitted to the Director of the Region II Office of Inspection and Enforcement within the time period specified for each report. These reports shall be submitted covering the activities identified in Table 6.9-1.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Reportable Occurrence Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of audits of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PRB and the SRB.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 RESPIRATORY PROTECTION PROGRAM

6.12.1 Allowance

Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table 1, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table 1, Column 1, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table 1, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in 20.101. These materials shall be subject to applicable process and other engineering controls.

6.12.2 Protection Program

In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in 6.12.1 above are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table 1, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or phychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair and storage.

5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee shall use equipment approved by the U.S. Bureau of Mines[3] under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U.S. Bureau of Mines[3] Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines[3] approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

6.12.3 Revocation

The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative con trol of the shift supervisor on duty.

3/and/or The National Institute of Health and Occupational Safety.

Table 6.2-1

Shift Manning Chart for Unit Operation

Staff Positions	AEC License Type Required (a)	Number of Personnel Required per Shift (b) (c) (d) (e) (f) (g)	
		Unit in Cold Shutdown Condition	Unit not in Cold Shutdown Condition
Shift Supervisor	SRO	1	1
Plant Operator	RO	1 (g)	1
Assistant Plant Operator	RO	1 (g)	1
Auxiliary Equipment Operator	N	0	2

Lower case letters in parentheses refer to the following notes.

NOTES:

a. The abbreviations used are:

SRO - Senior Reactor Operator
RO - Reactor Operator
N - Not Licensed

b. A licensed operator shall be in the control room at all times when there is fuel in the reactor. A Senior Reactor Operator shall be on-site at all times when there is fuel in the reactor.

c. Two licensed operators shall be in the control room during start-up, sheduled shutdown and during recovery from trips caused by transients or emergencies.

d. One member of each shift crew shall be qualified and designated to implement radiation control procedures.

e. A Senior Reactor Operator with no other concurrent operational duties shall directly supervise irradiated fuel handling and transfer activities and all fuel assembly transfers to or from the reactor vessel.

f. In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury, a qualified replacement shall be designated to report onsite immediately.

g. When the unit is in the Cold Shutdown Condition, it will not be necessary to have both a Plant Operator and and Assistant Plant Operator in the control room. Since both of these positions require a Reactor Operator's license either of the positions can handle the Cold Shutdown Condition requirements.

Table 6.9-1

UNIQUE REPORTING REQUIREMENTS

	Area	Tech Spec Reference	Submittal Date
a.	Primary Containment Leak Rate Tests (1)	4.7.A	Within 3 months following conduct of test
b.	Secondary Containment Leak Rate Tests (2)	4.7.C	Within 3 months following conduct of test
c.	Primary Coolant Leakage to Drywell	4.6.C	5 years (3)
d.	In-Service Inspection Evaluation	4-6.K	5 years (3)
e.	Reactor Coolant Radioactivity in excess of specified limits	4.6.F	Within 30 days of the occurrence

Notes:

1. Each integrated leak rate test of the primary containment shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report as described in the 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors", shall include data, analysis, and interpretations of the results which demonstrate compliance in meeting the specified leak rate limits.

2. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

3. The report shall be submitted within the period of time listed based on the commercial service date as the starting point.

TABLE 6.12-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES[1]	PROTECTION FACTORS[2] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES[3] APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask[4,7]	NP	5	21B 30 CFR § 14.4(b)(4)
Facepiece, full[7]	NP	100	21B 30 CFR § 14.4(b)(5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR § 12.2(c)(2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR § 12.2(c)(2) Type C(i)
Facepiece, full[7]	D	100	19B 30 CFR § 12.2(c)(2) Type C(ii)
Facepiece, full	PD	1.000	19B 30 CFR § 12.2(c)(2) Type C(iii).
Hood	CF	5	6
Suit	CF	5	6
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full[7]	D	100	13E 30 CFR § 11.4(b)(2)(i)
Facepiece, full	PD	1,000	13E 30 CFR § 11.4(b)(2)(ii)
Facepiece, full	R	100	13E 30 CFR § 11.4(b)(1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR § 12.2(e) or applicable schedules as listed above

1, 2, 3, 4, 5, 6, 7 (These notes are on the following pages)

3/and/or National Institute for Occupational

TABLE 6.12-1 (Continued)

[1]See the following symbols:

- CF: continuous flow
- D: demand
- NP: negative pressure (i.e., negative phase during inhalation)
- PD: pressure demand (i.e., always positive pressure)
- R: recirculating (closed circuit)

[2](a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency (above 99.9% removal efficiency by U.S. Bureau of Mines 3/ type dioctyl phthalate - DOP - test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

[3]Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote (5), below, concerning supplied-air suits and hoods.

[4]Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table 1, Column 1 of 10 CFR Part 20.

3/and/or National Institute for Occupational Safety and Health (NIOSH).

TABLE 6.12-1 (Continued)

[5]Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

[6]No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

[7]Only for shaven faces.

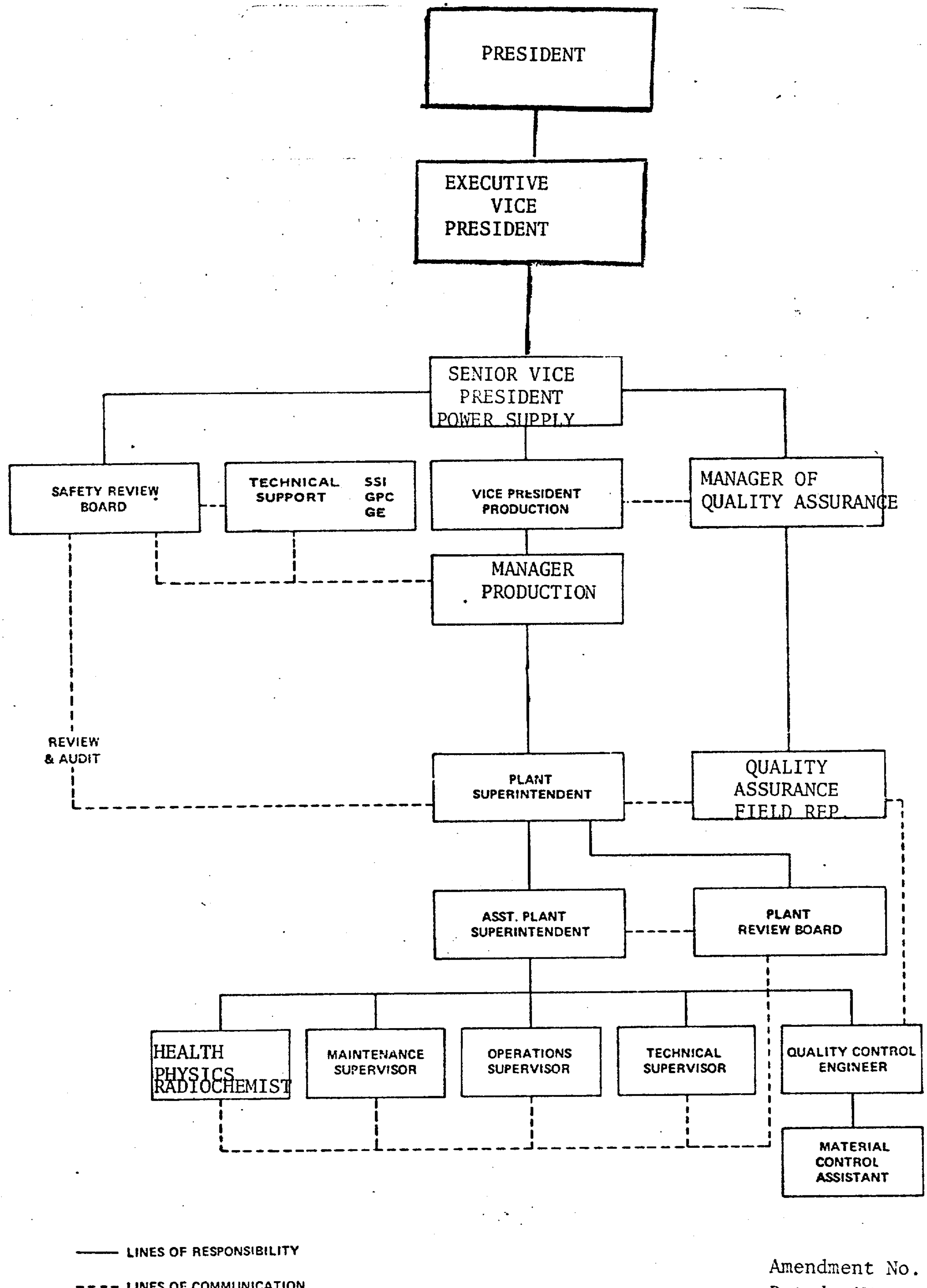
NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Burea of Mines[3] in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table 1 of this part are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

[3]and/or National Institute for Occupational Safety and Health (NIOSH).

FIGURE 6.2-1

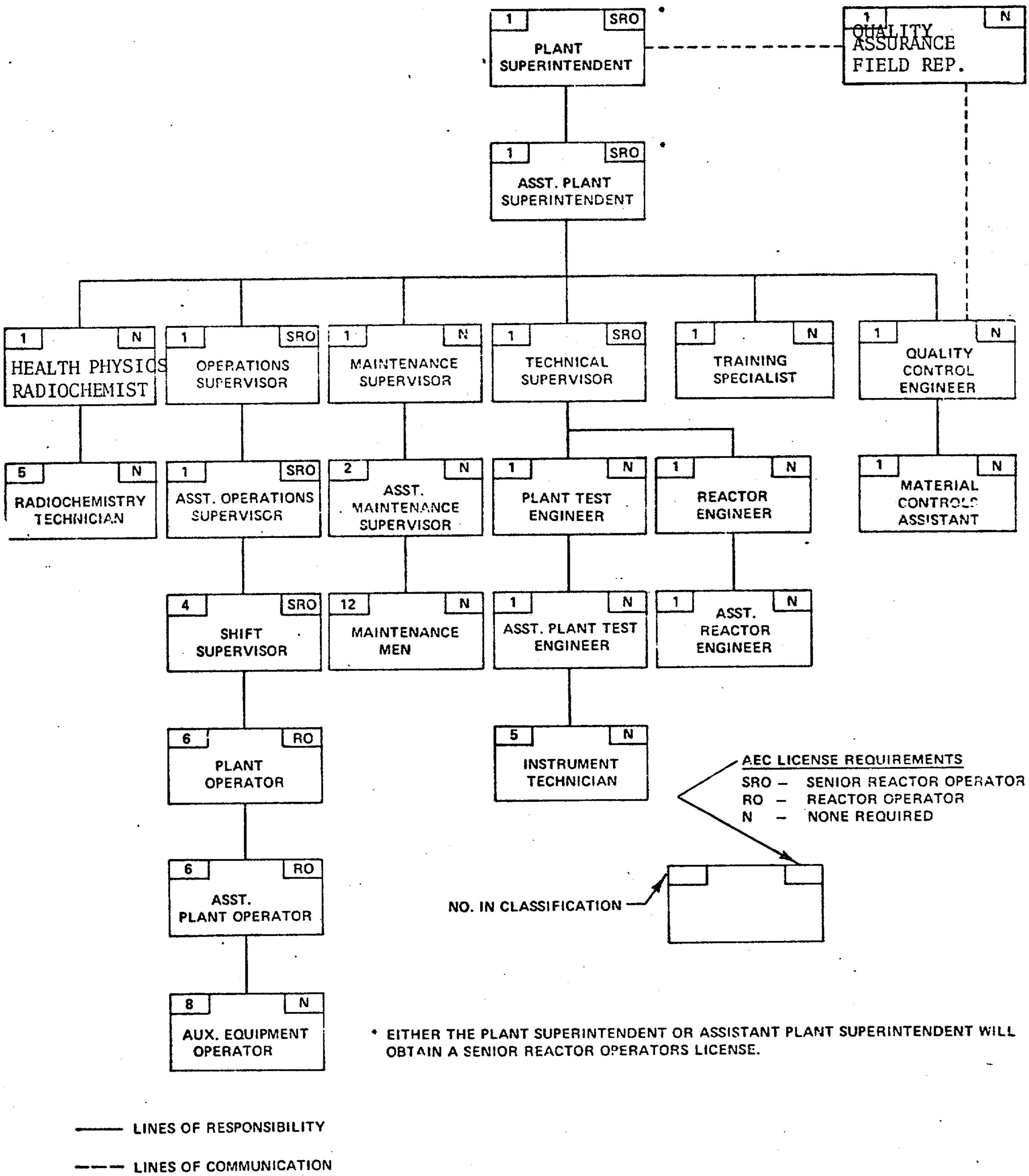
MANAGERIAL ORGANIZATION



Amendment No. 18
Dated: JAN 1 5 1975

FIGURE 6.2-2

PLANT ORGANIZATION



* EITHER THE PLANT SUPERINTENDENT OR ASSISTANT PLANT SUPERINTENDENT WILL OBTAIN A SENIOR REACTOR OPERATORS LICENSE.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO FACILITY LICENSE NO. DPR-57

GEORGIA POWER COMPANY AND OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-321

Introduction

By letter dated November 29, 1974, Georgia Power Company (GPC) proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-57, for the Edwin I. Hatch Nuclear Plant Unit 1. The proposed changes involve changes to the administrative controls including changes to the reporting requirements.

Discussiôn

The proposed changes would be administrative in nature and would affect the conduct of operation. The proposed changes are intended to provide uniform license requirements. Areas covered by the proposed uniform specifications include licensee staffing qualifications and management procedures involved with operating the reactor, reporting requirements, abnormal occurrence definition change, and a respiratory protection program.

Members of the facility staff should meet the requirements set forth in Guide 1.8, "Personnel Selection and Training" which endorses proposed ANSI N18.1, which was subsequently issued as ANSI N18.1-1971. Provisions for independent review of facility operations should be in accord with Guide 1.33, "Quality Assurance Program Requirements" which endorses proposed standard ANS 3.2, which was subsequently issued as ANSI 18.7-1972.

In Section 208 of the Energy Reorganization Act of 1974 "abnormal occurrences" is defined as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. The term "abnormal occurrence" is reserved for usage by NRC. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, enumerates required reports consistent with Section 208. The proposed change to required reports identifies the reports required of all licensees not already identified by the regulations and those unique to this facility. The proposal would formalize present reporting and would delete any reports no longer needed for assessment of safety related activities. In addition, a radiation protection program delineates use of respiratory equipment in the event personnel are to be exposed to concentrations in excess of Part 20 concentrations.

Evaluation

The new guidance for reporting operating information does not identify any event as an "abnormal occurrence". The proposed reporting requirements also delete reporting of information no longer required and duplication of reported information. The standardization of required reports and desired format for the information will permit more rapid recognition of potential problems.

Identifying minimum acceptable qualifications for facility personnel should assure capable performance from the facility staff. Other administrative requirements also restated by the specifications assure uniformity and conformance to the desired features in the review, staffing, and procedures. Incorporating the currently accepted respiratory protection program at this time assures that a consistent method of using respiratory equipment is immediately available whenever needed. Similar changes are being approved for all power reactor licensees, so all licensees will have the same requirements presented in a uniform manner.

During our review of the proposed changes, we found that certain modifications to the proposal were necessary to have conformance with the desired regulatory position. These changes were discussed with your staff and have been incorporated into the proposal.

We have concluded that the proposal as modified improves the licensee's program for evaluating plant performance and the reporting of the operating information needed by the Commission to assess safety related activities and is acceptable. The facility staff qualifications and training program conform to Guide 1.8 and therefore are acceptable. The administrative procedures and facility review and audit are consistent with Guide 1.33 and are acceptable. The modified reporting program is consistent with the guidance provided by Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4. The administrative controls are consistent with requirements being incorporated in Technical Specifications for new licensed facilities.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 15, 1976

or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 29, 1974, (2) Amendment No. 18 to License No. DPR-57, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Appling County Public Library, Parker Street, Baxley, Georgia 31513.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 15th day of January, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

OFFICE➤	ORB#3	ORB#3	ORB#3			
SURNAME➤	CParrish:kmf	JGuibert	GLear			
DATE➤	1/ 16 /76	1/ 16 /76	1/ 16 /76			

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY AND
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. DPR-57 issued to Georgia Power Company and Oglethorpe Electric Membership Corporation (the licensees) which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant Unit 1, located in Appling County, Georgia. The amendment is effective 60 days from the date of issuance.

The amendment incorporate changes related to Administrative Controls into the Technical Specifications for Edwin I. Hatch Nuclear Plant Unit 1.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration